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Feasibility Study for Applicability of Ageing PSA Model Results in Risk-informed Decision Process

*APSA Network on Use of PSA
for Evaluation of Ageing
Effects - Task 8*

APSA



Shahen Poghosyan,
Mirela Nitoi and
Zdenko Šimić

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Institute for Energy and Transport

Contact information

Ghislain Pascal
Address: Joint Research Centre, PO Box 2, NL-1755 ZG Petten
E-mail: ghislain.pascal@ec.europa.eu
Tel.: +31 224 56 56 56
Fax: +31 224 56 56 37

<http://www.jrc.ec.europa.eu/>

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Feasibility study for applicability of ageing PSA model results in risk-informed decision process

APSA Network Task 8

Shahen Poghosyan, Mirela Nitoi and Zdenko Šimić

Abstract

The aim of this case study was to demonstrate the benefits of using ageing probabilistic safety analysis (PSA) results in risk-informed decision process.

First stage of the activities was aimed on the development of dataset which reflects age-dependent reliability behaviour for selected components using plant-specific, VVER specific and generic data. The second part describes the integration process of time-dependent reliability dataset in PSA model, quantification and comparison between aging (APSA) and base case PSA decision making results. Practical insights and conclusions are also presented.

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1. INTRODUCTION

Core damage frequency (CDF), risk profile and risk Importance factors, which are being used as decision criteria in many applications, could evolve during the lifetime of the plant, and these changes have to be considered in decision making process. This feasibility study is aimed to demonstrate the benefits of using ageing probabilistic safety assessment (PSA) results in risk-informed decision process.

The work was performed within Task 8 of Ageing PSA European Network (APSA, [1]), and it was dedicated to practical applications that use the addressing of ageing effects in PSA, in order to evaluate the impact of ageing on the overall plant safety. Applications could be related to:

- prioritization of safety-related activities using APSA results,
- predictive evaluation of plant safety level,
- prioritization of ageing management issues, LTO (long term operation) activities using APSA findings.

First stage of the activities was aimed on development of dataset which reflects age-dependent reliability behaviour for selected components using plant-specific, VVER specific and generic data. The second part describes the integration process of time-dependent reliability dataset in PSA model, quantification and comparison between APSA and base case PSA decision making results.

2. TASK SPECIFICATION

2.1 SELECTION OF COMPONENTS

Selection of components was aimed to reveal those which:

- have significant role for nuclear safety and
- have the potential to be seriously affected by ageing effects.

Complementary techniques have been used for component selection, in order to assure the completeness of components list for further consideration.

The components were selected based on a combination of the following criteria [4]:

- numerical risk importance criteria (to address safety significance issue),
- qualitative judgment (AFMEA - Ageing Failure Mode and Effects Analysis technique, to address ageing vulnerability issue).

In order to optimize the available resources, estimation of time-dependency of reliability parameters has been performed only for selected components.

Main approach for component selection according to safety significance was the development of risk-based components list, using risk parameters, based on probabilistic model available for Armenian NPP Unit 2. To determine the risk-importance of safety systems and components, the commonly used risk importance measures are the Fussell-Vesely (FV) and Risk Increase Factor (RIF). Fussell-Vesely importance factor is calculating by the following formula:

$$FV = \frac{\sum_{j=1}^c S_j}{Q} \quad (1)$$

where:

- S_j – frequency of minimal set of failures which could lead to core damage,
- c – number of minimal cut-sets which contains basic events with failures of particular equipment or human error,
- Q – total frequency of core damage event.

Physical meaning of this expression is proportion of minimal cut-sets with particular failure and total frequency of core damage event

Limitation of Fussell-Vesely importance factor resides in neglecting the components which have low failure probability. However, since the failure probability may rapidly increase due to ageing, the use of Fussell-Vesely importance factor is not enough to assure the completeness of components list. During discussions with APSA Network partners, it was suggested to use also the risk increase factor (RIF) as an indicator, calculated by:

$$RIF = \frac{Q_{P=1}}{Q} \quad (2)$$

where $Q_{P=1}$ – conditional core damage frequency with particular failure.

Use of risk increase factor allows taking into account the components with low failure probability but large consequences, in regard with core damage.

Screening criteria were established at 0.5 % ($FV > 0,005$), as it was recommended by US NRC [6] and EPRI [7] and at 100 for RIF.

The main problem of risk-based analysis is lack of information regarding equipment which is not in PSA model, but potentially could have a significant importance due to ageing-based decrease

of reliability parameters. To solve this issue the ageing failure mode and effects analysis (AFMEA) procedure was used, [5].

Application of the above described approach leads to the following list of components, which were analyzed in detail (see Table 1).

Table 1 - List of components for further consideration

#	Components	Base for components selection			considered failure modes	component boundaries,
		FV	RIF	AFMEA		
1.	Main feedwater pumps	+			Fail to run Fail to start	Flow – welding and flanges in pipe Power supply – connection of cable to pump breaker (breaker is within pump's boundaries)
2.	Emergency make-up pump	+	+			
3.	Spray pumps	+	+			
4.	RHR pumps	+				
5.	Intermediate circuit pump	+				
6.	Emergency feedwater pump	+				
7.	Drainage pumps	+				
8.	Emergency seismic feedwater pump	+				
9.	Normal make-up pump	+				
10.	Check valve (D<100mm)			+	Fail to open Fail to close	Flow – welding and flanges in pipe
11.	Check valve (D≥100mm)			+	Fail to open Fail to close Fail to remain close Fail to remain open	
12.	Motor valve (D<100mm)	+	+		Fail to change position	Flow – welding and flanges in pipe Power supply – connection of cable to pump breaker (breaker is within component's boundaries)
13.	Motor valve (D≥100mm)	+	+			
14.	Motor ducts	+				
15.	Manual valve	+	+		Fail to change position Fail to remain open Fail to remain close	Flow – welding and flanges in pipe Power supply – connection of cable to pump breaker (breaker is within component's boundaries)
16.	Steam-dump valves	+			Fail to change position Fail to remain close	Flow – welding and flanges in pipe Power supply – connection of cable to pump breaker (breaker is within component's boundaries)
17.	TG stop valves	+			Fail to close	Flow – welding and flanges in pipe Power supply – connection

#	Components	Base for components selection			considered failure modes	component boundaries,
		FV	RIF	AFMEA		
						of cable to pump breaker (breaker is within component's boundaries)
18.	Fast changing filters	+			Fail to operate	Flow – welding and flanges in pipe
19.	Diesel-generator	+			Fail to run Fail to start	Power output - connection of cable to generator Service water – valve on cooling line (valve is not within component's boundaries) Fuel – welding on pipe between external and internal fuel system Oil – welding on pipe between external and internal oil system
20.	Switchers	+			Fail to operate	Commutable lines – connection of cable to the key
21.	Interlocks switchers	+				Power supply - connection of cable to cubicle's breaker (breaker is not within component's boundaries)
22.	0.4kV buses (cubicle)	+	+			Flow – connection to the impulse line Power supply - connection of cable to Transducer Control signals – inlet and outlet of control cables of the device
23.	Level sensor		+			Power supply - connection of cable to Transducer Control signals – inlet and outlet of control cables of the device
24.	Pressure sensors'		+			Power supply - connection of cable to Transducer Control signals – inlet and outlet of control cables of the device
25.	Transducer of level sensors'		+			Power supply - connection of cable to Transducer Control signals – inlet and outlet of control cables of the device
26.	Transducer of pressure sensors'		+			Power supply - connection of cable to Transducer Control signals – inlet and outlet of control cables of the device
27.	Transformer ≤ 6kV		+			Power supply - connection of cables to the transformer Power supply - connection of cables to electrical bus
28.	6kV buses	+	+		Fail to change position	Commutable lines – connection of cable to the key Control signals – connection of control cables of the breaker
29.	0.4kV breakers	+	+			
30.	6kV breakers	+	+			

Selected components failure statistics has been thoroughly examined, in order to identify any increasing ageing trend, or to assure that constant failure rate model is applicable.

2.2 DATA SOURCES

Development of time-dependent reliability parameters dataset has been initiated by data gathering process for selected components. The following data sources have been used:

- plant-specific data,
- data from similar NPP,
- generic sources.

TIRGALEX database [10] was mainly used for active components ageing trend evaluation and NUREG-1829 [11] was used for estimation of LOCA frequencies.

Plant-specific data

Plant-specific data have been collected, from Armenian NPP Unit 2 systems and components failure records. Mainly PSA database was used since all of the selected components are modelled in PSA. Particularly, data contains the following information:

- Component name,
- System name,
- Component type (pump, valve, etc.),
- Failure impact,
- Character of failure,
- Date and timing,
- NPP operational mode,
- Root cause.

Collected data are covering the period of plant operation starting from 1995 to 2007. Since plant-specific operational data from one unit is not sufficient for comprehensive time-dependent reliability analysis, the decision has been taken to use generic data information.

Data from similar VVER-440 units

Statistical data gathering process has been established in the frame of case study performed for VVER-440 equipment [9]. As a first step, the detailed technical description of selected components was distributed between Ageing PSA European Network participants. Data on selected components was gathered from similar VVER-440 plants (operating in Czech Republic, Hungary, Slovakia) and it was provided only for the approximately similar components (from point of view of technical parameters, functions, operating conditions etc.). Operational data have been gathered from 14 units of NPP which are currently in operation.

Detailed information on design and reliability characteristics and registered failures was presented in the following structure:

- Component description,
- Commissioning date,
- Component boundaries,
- Observation period / covered ages window,
- Considered failure modes and operating states,
- # of components in the sample for data collection,
- # of observed failures per failure mode (# of observed non-critical failures),
- Dates of failures / failure mode / component ID (Cause of failure),
- Average # of hrs/demands per year per population,
- Important modifications (is there any significant measures taken?),
- Performance characteristics,
- Environmental conditions,
- Period of maintenance.

Availability of this kind of representative data enables confident interpretation of results.

2.3 TIME-DEPENDENT RELIABILITY DATASET

Two stages of data processing have been applied, [12]:

1) Application of non-parametric test

Non-parametric methods allow identifying whether the ageing trend is present, for a particular component. However, non-parametric methods are not able to give information about pattern of the trend. The test considers the null hypothesis (H_0) which states that failure rate values (λ_i) are identically distributed against the alternative that a trend is present.

2) Application of parametric test

The aim of parametric methods application is to identify a pattern for components ageing trend. Parametric methods were applied for cases when non-parametric analysis rejects H_0 hypothesis.

The following statistical models were applied for reliability parameter function ($p(t)$ - probability of failure on demand, θ_1, θ_2 - parameters, t – time):

- Constant failure rate model,
- Log-linear model: $\ln \varphi(\vec{\theta}; t) = \theta_1 + \theta_2 t$
- Power-low or Weibull model: $\lambda(\tau) = \varphi(\vec{\theta}; t) = \theta_1 t^{\theta_2}$
- Complementary log-log model: $\lambda v(-\lambda v(1 - \pi(\tau))) = \theta_1 + \theta_2 \tau$,
- Logit model: $\ln(p(t)/(1 - p(t))) = \theta_1 + \theta_2 \tau$,
- Probit model: $\Phi^{-1}(\pi(\tau)) = \theta_1 + \theta_2 \tau$, where $\Phi^{-1}(\pi(\tau))$ - is the inverse function of cumulative normal distribution.

Checking of reliability models assumptions was implemented using p-value “best fitting” criteria ($p=0,5$), [12]. According to the results of non-parametric test, application H_0 hypothesis was considered appropriate for the following equipments:

1. Secondary device of pressure sensors,
2. Emergency make-up pump,
3. Drainage pumps,
4. RHR pumps,
5. Main feedwater pumps,
6. Diesel-generator,
7. Normal make-up pump,
8. Fast changing filters,
9. Primary device of level sensors,
10. Secondary device of level sensors.

For the above listed equipments, the constant failure rate reliability model is applicable.

For cases when H_0 hypothesis has been rejected, parametric test was applied to identify the pattern of components ageing trend. The results of parametric test application (results derived using WinBUGS code) are presented in Table 2.

Table 2 - List of components with increasing ageing trend observed

#	Component	Failure mode	Pattern	θ_1	θ_2
1.	6kV buses	Fail to operate	Loglinear	-13.01	0.08314
2.	Switcher	Fail to operate	Loglinear	-15.39	0.3145
3.	0.4kV breaker	Fail to change position	Power	2.298	4.12E-07
		Fail to remain in position	Loglinear	-16.74	0.1251
4.	Motor valve (D>100mm)	Fail to change position	Loglinear	-13.65	0.07406
5.	Pressure sensor	Fail to operate	Loglinear	-18.41	0.236
6.	6kV breaker	Fail to change position	Loglinear	-13.23	0.0284
7.	Motor valve (D<100mm)	Fail to change position	Power	4.79E-06	0.346

Plant-specific or VVER specific data were not available for the rest of selected components, so generic TIRGALEX ageing trend data was assigned (see Table 3).

Table 3 - Generic ageing rate values of components with lack of VVER-specific information

#	Component	Failure mode	Ageing rate [h ⁻¹ y ⁻¹]	Base case value ¹
1.	0.4kV bus	Fail to operate	1,10E-09	4,02E-07
2.	Manual valve	Fail to change position	2,73E-06	8,69E-05
		Fail to remain open	2,20E-09	5,86E-07
		Fail to remain close	2,20E-09	7,01E-08
3.	Intermediate circuit cooling pump	Failure on demand	2,20E-07	3,12E-05
		Fail to run	2,20E-07	1,23E-04
4.	EFW and ESW	Failure on demand	2,20E-07	3,87E-06
		Fail to run	2,20E-07	9,61E-05
5.	Transformer ≤ 6kV	Fail to operate	1,70E-09	7,25E-07
6.	Check valve (D≤100mm)	Fail to open	3,80E-09	8,14E-08
		Fail to close	3,80E-09	7,32E-07
7.	Check valve (D>100mm)	Fail to open	3,80E-09	2,62E-07
		Fail to close	3,80E-09	9,30E-08
		Fail to remain closed	3,80E-09	1,88E-07
		Fail to remain open	3,80E-09	9,43E-08
8.	Steam dump valves	Fail to change position	2.17E-07	5.28E-06
		Fail to remain closed	7,00E-07	1,70E-05
9.	Spray pump	Failure on demand	5.31E-09	5.00E-06
		Fail to run	2.20E-07	2.07E-04
10.	Interlock switch	Fail to operate	2.30E-07	1.08E-06
11.	Motor driven duct	Fail to change position	3.60E-06	2.64E-5
12.	TG stop valves	Fail to close	4.00E-07	8.46E-06

Beside the above components, primary piping ageing trends have been revised using investigation described in NUREG-1829, [11].

¹ value used in PSA model

2.4 INCORPORATION OF T-D RELIABILITY DATASET IN PSA MODEL

Ageing PSA model has been developed based on third iteration of base case level 1 PSA model for Armenian NPP Unit 2. The characteristics of the model are as follows:

- *Undesired event considered*: Damage of the fuel located in the reactor core
- *Considered regimes*: 50-100% of nominal power (regimes with both turbines in operation)
- *Considered initiators*: Internal initiating events

Base case PSA model corresponds to 25 years of operation and it was developed using Risk Spectrum code. Integration process of time-dependent reliability dataset was implemented by Risk Spectrum «Exchange event» function. Parameters of basic events in the model were changing against boundary conditions, which represent different plant lifetime periods (four periods were considered -30 years, 35 years, 40 years and 45 years).

Time-dependent reliability dataset was developed within stage 1 of feasibility study and it contains the list of components considered in the analysis, divided in three main parts:

- 1) components for which specific pattern of increasing ageing trends were revealed,
- 2) components for which generic linear ageing trends were conservatively applied due to lack of statistical data² and
- 3) components for which constant reliability model was found.

For the rest of selected component groups applicability of constant failure mode was justified within stage 1.

2.5 PSA MODELS QUANTIFICATION

Quantification was performed for base case PSA model and PSA model with integrated time-dependent reliability dataset (Ageing PSA model). The aim of quantification process is to calculate the frequency of core damage and to derive risk profile information. Risk Spectrum quantification module gave also information about risk importance factors, sensitivity parameters and uncertainty values.

Two stage quantification processes was applied. Post-processing has been used In order to cover human errors dependency. Results presented below are obtained after post-processing calculations. Core damage frequency (CDF) calculations were done taking into account minimal cut-set analysis basic parameters with absolute cut-off value 10^{-14} [1/y].

Base case PSA model quantification

Core damage frequency analysis show that CDF mean value is equal to $8.50\text{E-}05$ [1/y], with upper bound equals to $1.36\text{E-}04$ [1/y] and lower bound equals to $5.30\text{E-}05$ [1/y]. Risk profile obtained during quantification is presented on Figure 1.

As it is shown in Figure 1, 80% of total CDF value is conditioned by the first 9 initiators. The role of different mitigation measures failure for mentioned initiators is presented in Appendix 2.

Figure 2 presents the contribution to overall CDF coming from different initiating event classes. Classification of initiating events is performed based on IE consequences and plant responses similarity. It could be easily noticed from Figure 2 that the largest core damage risk contributors are primary and secondary leaks, which contributes with 83% of overall core damage frequency.

² Preliminary quantification was performed without generic ageing trend applied for the components with lack of statistical data, and the results obtained were compared with final quantification results. Comparison shows that results are mainly dominated by increasing ageing trends, revealed using plant-specific statistics. To assure consistency, it was decided to use integrated ageing PSA model with generic ageing trends applied for these components, since impact of those data is negligible for decision making process.

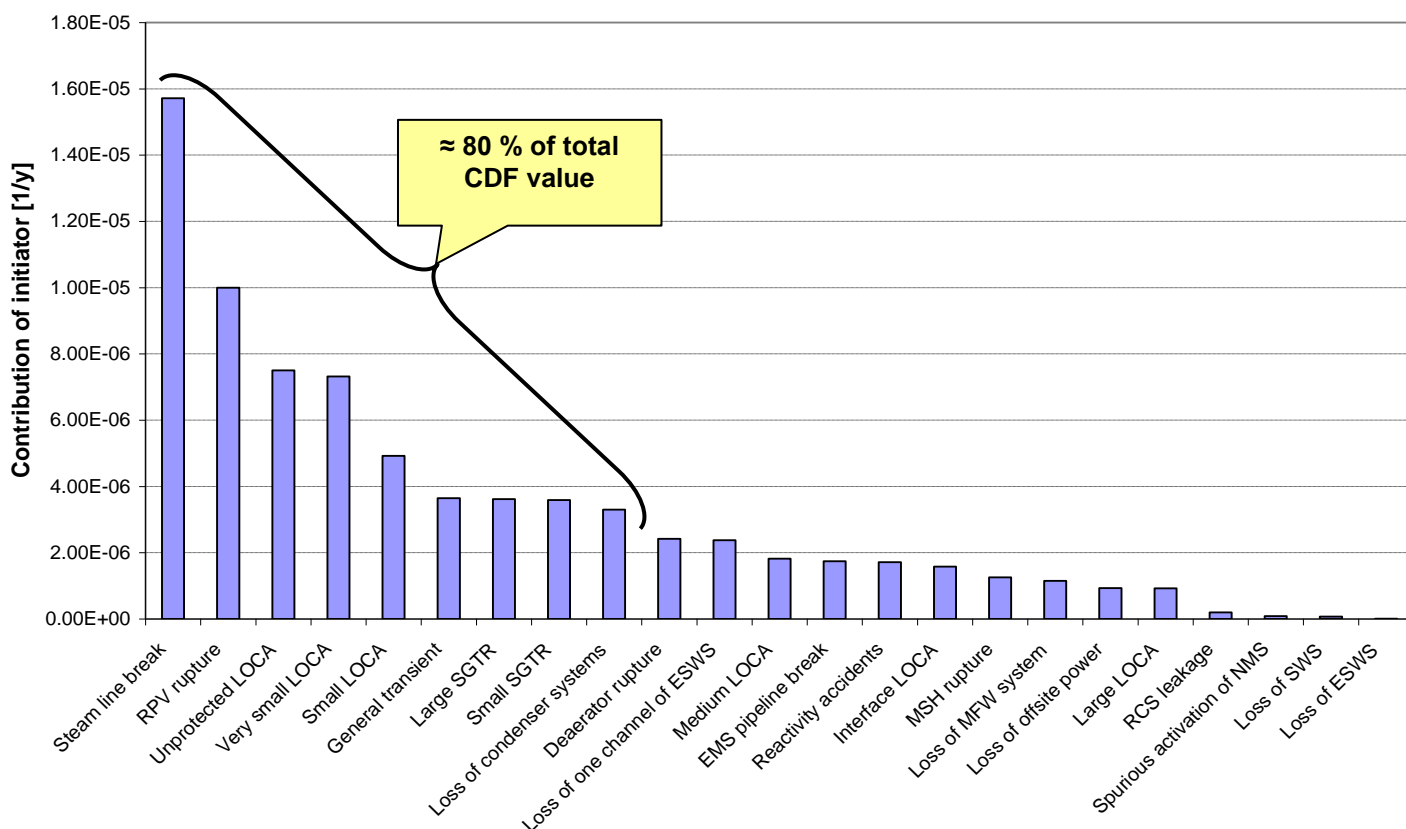


Figure 1 - Risk profile for base case PSA model (25 years of operation)

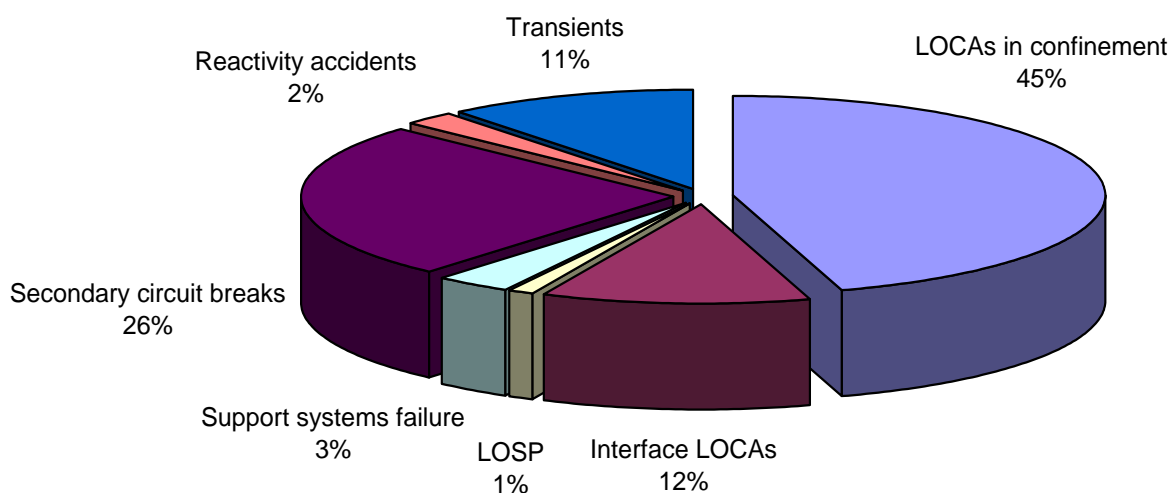


Figure 2 - Contribution of different IEs classes for base case PSA model (25 years of operation)

Primary leaks

Minimal cut-sets of LOCA events are dominated by several factors connected with failure of mitigation systems: emergency core cooling system (ECCS), spray system (SS), essential loads service water system (ESWS), AC power supply system, DC power supply system, etc. However, as it is shown in Appendix 2, the risk coming from LOCA events is mainly dominated by 2 groups of minimal cut-sets, related to:

- ECCS failure to provide water in primary circuit and
- failure of spray system to perform boron tank cooling function.

Those groups are changing their contribution portion depending on LOCA type: for small LOCAs main contribution is coming from spray system failures, for bigger ones – ECCS failures. This is conditioned by the fact that for small LOCAs the success criteria of ECCS is significantly changing, from small diameter LOCAs to the big ones, whereas for spray system boron tank cooling function success criteria remains the same. Overall picture could be summarized as follows: emergency core cooling system and spray system have similar role in LOCAs sector presented on Figure 2.

Secondary leaks

Minimal cut-sets of secondary leakage events are dominated by several factors, connected with failure of mitigation systems: steam leak isolation valve system, SG feedwater systems (secondary feed & bleed function), primary feed & bleed systems (ECCS + SS), AC power supply system, DC power supply system, etc. However, as it is shown in Appendix 2, the risk coming from secondary leakage events is mainly dominated by 3 groups of minimal cut-sets, related to:

- Failure to close fast steam isolation valves, to prevent reactor cold overpressure effect,
- ECCS failure to provide water in primary circuit, during primary feed & bleed procedure and
- Failure of spray system to perform boron tank cooling function, during primary feed & bleed procedure.

In case of failure of FSIV, the secondary side leakage will lead to rapid decrease of primary temperature, which will lead to cold overpressure. In case of successful leak isolation it is foreseen to implement primary feed & bleed (F&B) function. It is assumed that in case of steam line breaks, steam and humidity in turbine hall will lead to damages of equipment responsible for secondary feed & bleed function (steam dump valves, SG feedwater pumps, etc).

The specific cases are unprotected LOCA and reactor pressure vessel rupture events. Those events are directly leading to core damage. The only possible mitigating measure came from consideration of leak before break concept, which could significantly decrease probability of unprotected LOCA event.

Ageing PSA model quantification

Integrated APSA model has been quantified and calculated for each considered case (30 years, 35 years, 40 years and 45 years), using the following assumptions:

- a) no modernizations or significant replacement of equipment are foreseen,
- b) CCF models recalculated based on new reliability parameters values (see tables 5 and 6),
- c) human error probabilities are constant,
- d) maintenance unavailability values (U_M) are correlated with failure rate increasing, by applying new failure rate (λ) value in maintenance unavailability model

$$U_M = \lambda T_M$$

where T_M is average time for maintenance,

- e) error factors assigned to reliability parameters are not dependent from failure rate values.

Core damage frequency analysis results for ageing PSA model are presented in Table 4.

Table 4 - Results of CDF calculations for Ageing PSA model

#	CDF	Base case CDF [1/y]	CDF calculations with incorporated ageing aspects			
		25	30	35	40	45
1.	Mean value	8.50E-05	1.20E-04	2.12E-04	6.93E-04	1.18E-02
2.	Upper bound	1.36E-04	1.89E-04	3.48E-04	1.17E-03	2.72E-02
3.	Lower bound	5.30E-05	7.48E-05	1.26E-04	3.97E-04	6.03E-03

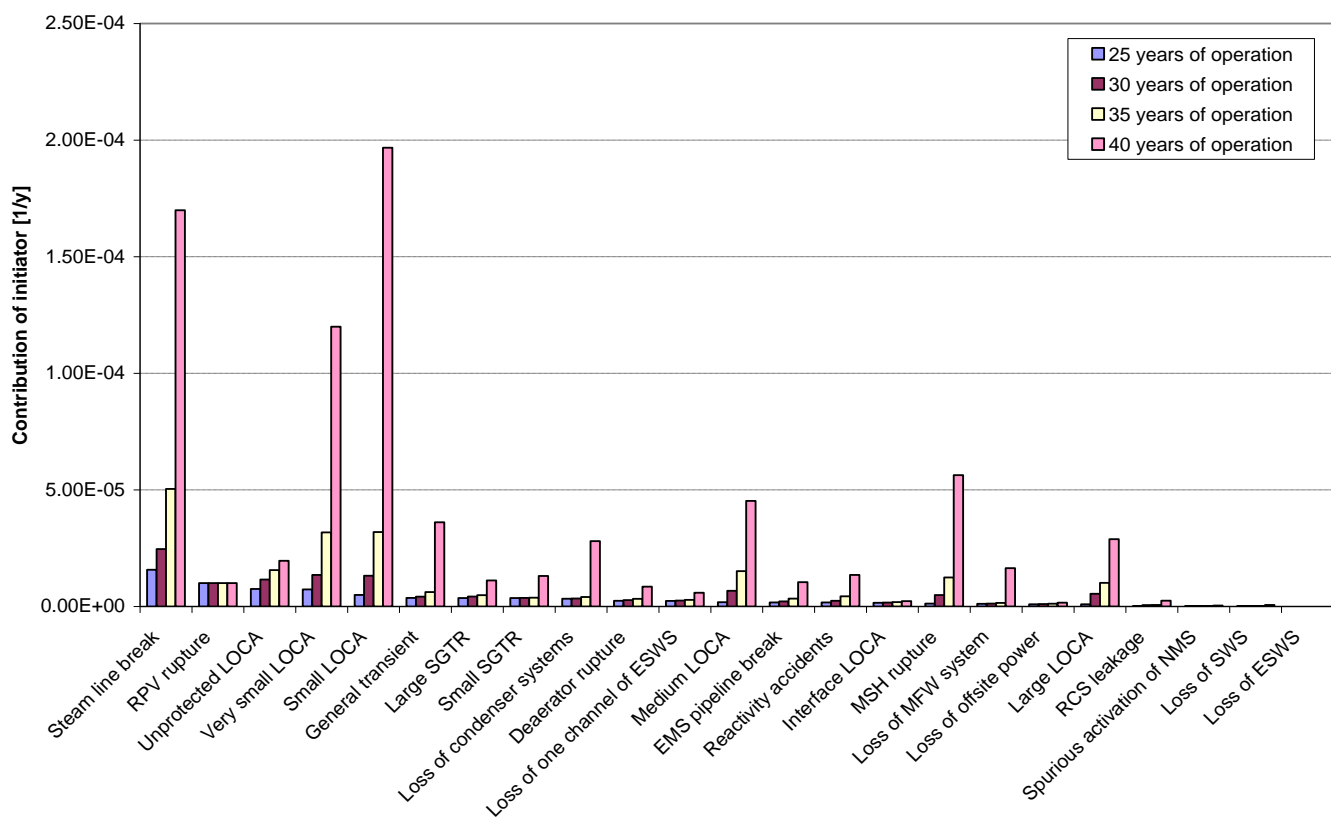


Figure 3 - Predictive risk profile behaviour depending on lifetime

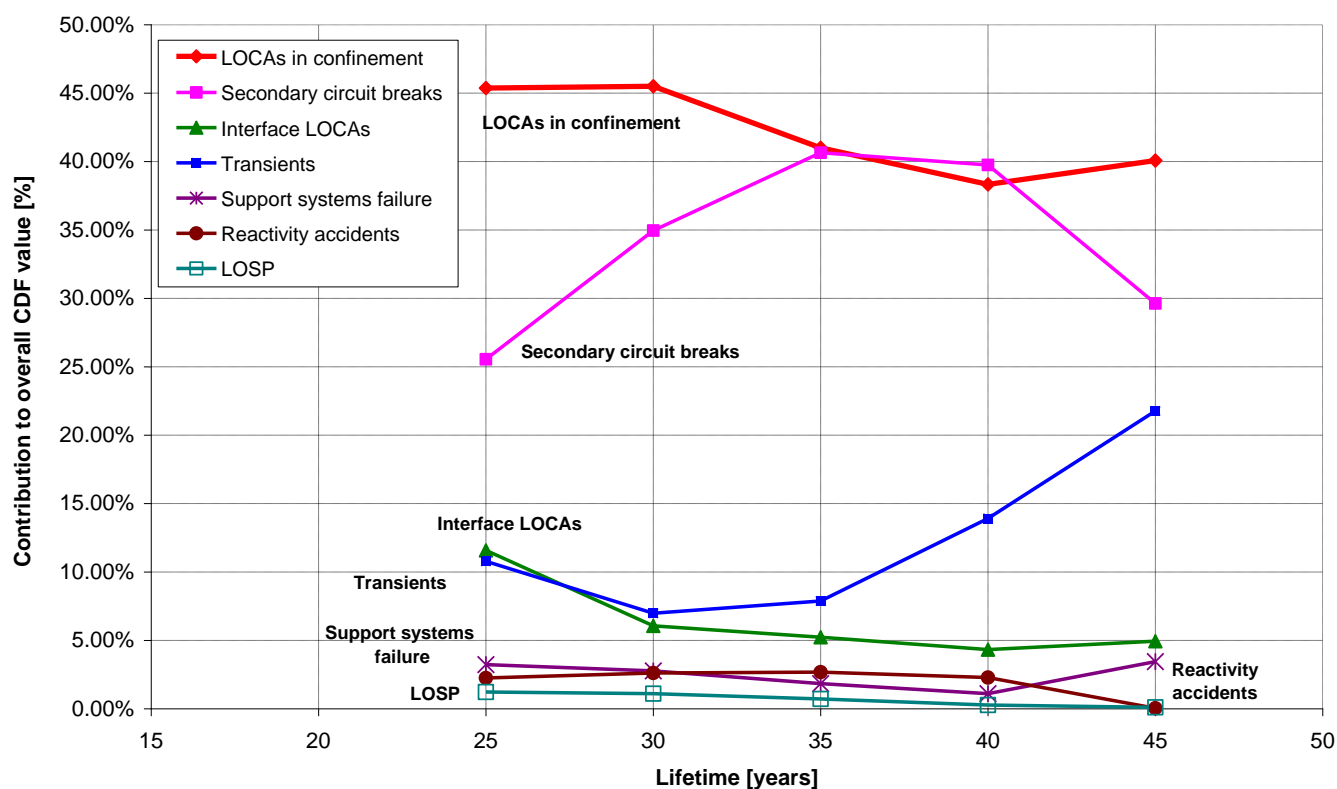


Figure 4 - Change of proportion from main risk contributors depending on lifetime

The results show a rapid increase of CDF value (by a factor of 120) in the period from 40 to 45 years of operation, while in the period from 25 to 40 years, CDF value increased only by a factor of 8.15. As it is presented in table 4, large uncertainties are found for long-term predictions (45 years operation), and based on that, it was concluded that adequate predictions could be done only for relatively short time periods (<15 years), which are comparable with the period for which statistical data are available. Hence, the results obtained for “45 years” case have not been taken into account for further comparison.

Risk profile obtained during quantification is presented on Figure 3.

Although events associated with primary and secondary leakages remain main contributors, the overall proportion of their contribution is decreased. At the same time the proportion of risk coming from transients is disproportionately increased (see Figures 3 and 4).

General transient (GT) implies occurrence of scram, without any additional failures in plant systems. Hence MCSs of GT contains multiple failures related to primary and secondary feed & bleed functions, which are significantly affected by ageing.

The detailed discussion of quantifications results is presented in chapter 3.

2.6 RISK-INFORMED DECISION MAKING PROCESS

Results of PSA models quantification assure an adequate base for decision making process, in terms of risk information. The Integrated Risk Informed Decision Making (IRIDM) process is a systematic way of taking into account all the relevant factors in making decisions on a wide range of safety, security or regulatory issues that could arise for any type of nuclear facility [2].

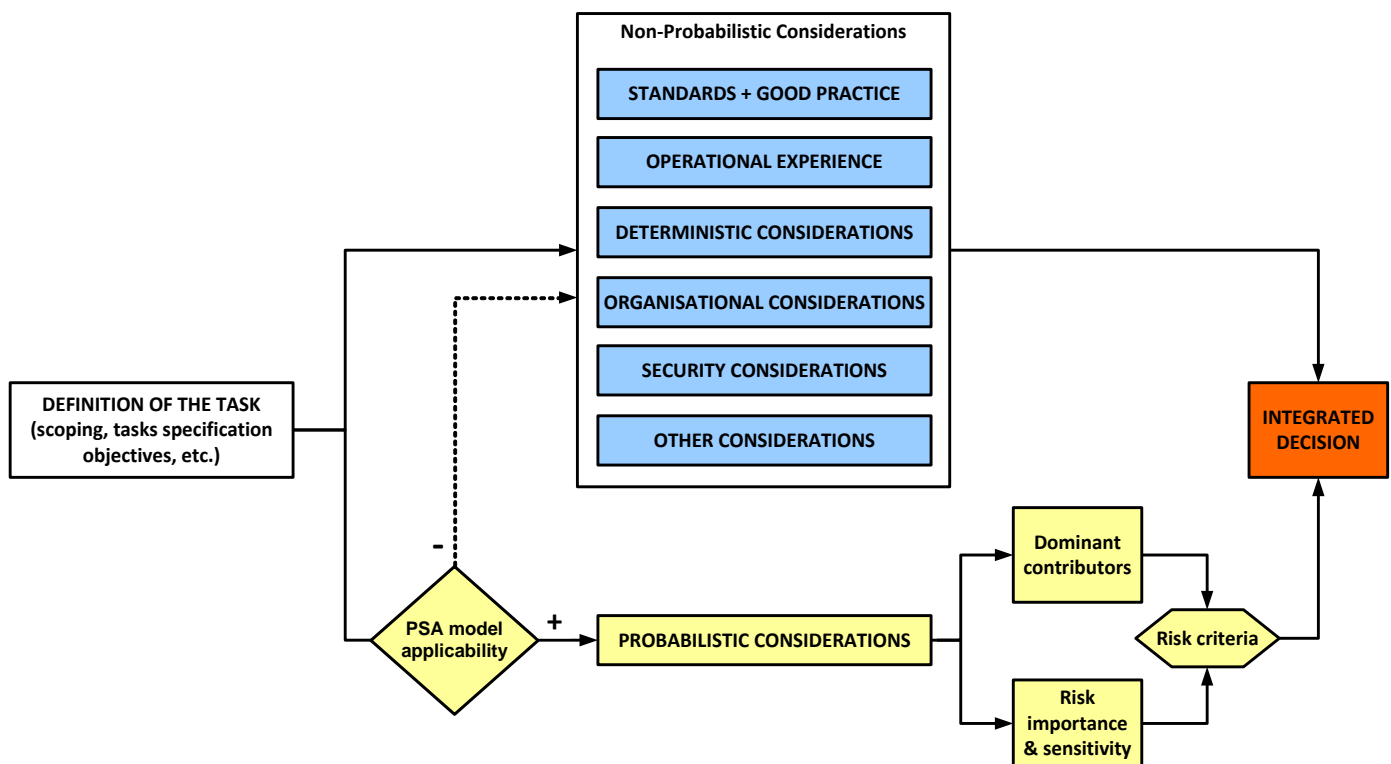


Figure 5 - Flowchart of integrated risk-informed decision making process

In applying the IRIDM process, all the relevant factors are identified and assessed to provide inputs to decision making process (see Figure 5). These factors typically include mandatory requirements (such as legal requirements and regulation), the insights from the deterministic analysis (such as defence in depth and safety margins), the insights from the risk assessment (usually obtained from a PSA model) and any other considerations that are relevant to the issue being addressed (such as radiation doses to workers and members of the public, or cost-benefit analysis). The inputs to the decision making process include the results and information from qualitative and quantitative analyses, [2].

The inputs to evaluate the options need to be established, and they can be presented by two main categories: non-probabilistic and probabilistic. Non-probabilistic inputs include the following:

- **Standards and good practices:** this relates to the standards and good practices recognised in the Member State and includes legally binding requirements, regulatory requirements, national and international standards produced by professional bodies, standards and codes recognised by the nuclear industry, managerial procedures, and good practices identified from similar facilities;
- **Operational experience:** this relates to the operating experience from the nuclear facility itself and for similar facilities relating to the issue being addressed and requires that a review of the operating experience is carried out;
- **Deterministic considerations:** this relates to the way that the basic deterministic principles have been addressed and includes the insights from: design basis accident analysis, analysis of defence in depth and safety margins;
- **Organisational considerations:** this relates to the management of safety aspects of the plant operation for the issue/ options being considered, including maintenance activities, training and plant procedures;
- **Security considerations:** this relates to the physical protection of the facility and requires that the interaction between safety and security measures is taken into account in addressing the issue/ options;
- **Other considerations:** this relates to the other requirements that may need to be addressed for specific issues/ options and includes: the radiation doses to workers in making plant changes; the costs and benefits from making plant modifications; and the remaining lifetime of the plant. This should also take into account the research being carried out, that relates to this issue.

Probabilistic considerations relate to the input from the PSA model, and include information on dominant contributors and risk significance. However it is also very important to check applicability of PSA model for a particular task analysis. It could be the case when the scope or level of resolution of PSA model could be insufficient, for a particular decision to be made. For example, full power PSA model could not be used for shutdown risk related evaluations. PSA applicability for a particular case could be verified using IAEA TECDOC-1511 [3]. Obtained PSA results should be compared with risk criteria (if any) in order to come-up with prescribed decision principles.

Integrated decision is made by weighting the above listed inputs. The importance of each input element is directly dependent upon the decision to be made [2].

3. RESULTS

Risk profile

Changes of risk profile and proportion coming from main risk contributors are conditioned by changes in mitigation functions failure probability. The distribution of contribution of mitigation functions is shown in Appendix 2. According to described results, CDF value is highly dependent on reliability of the following safety systems and their associated functions:

- ECCS to provide cooling water to primary circuit,
- Spray system to provide cooling of boron tank and
- Steam isolation valves to prevent PTS.

The importance of mentioned mitigation functions is conditioned by large contribution of primary and secondary leakage events³. Time behaviour of mitigation functions in overall MCS, for a specific initiator, is presented in Figure 6.

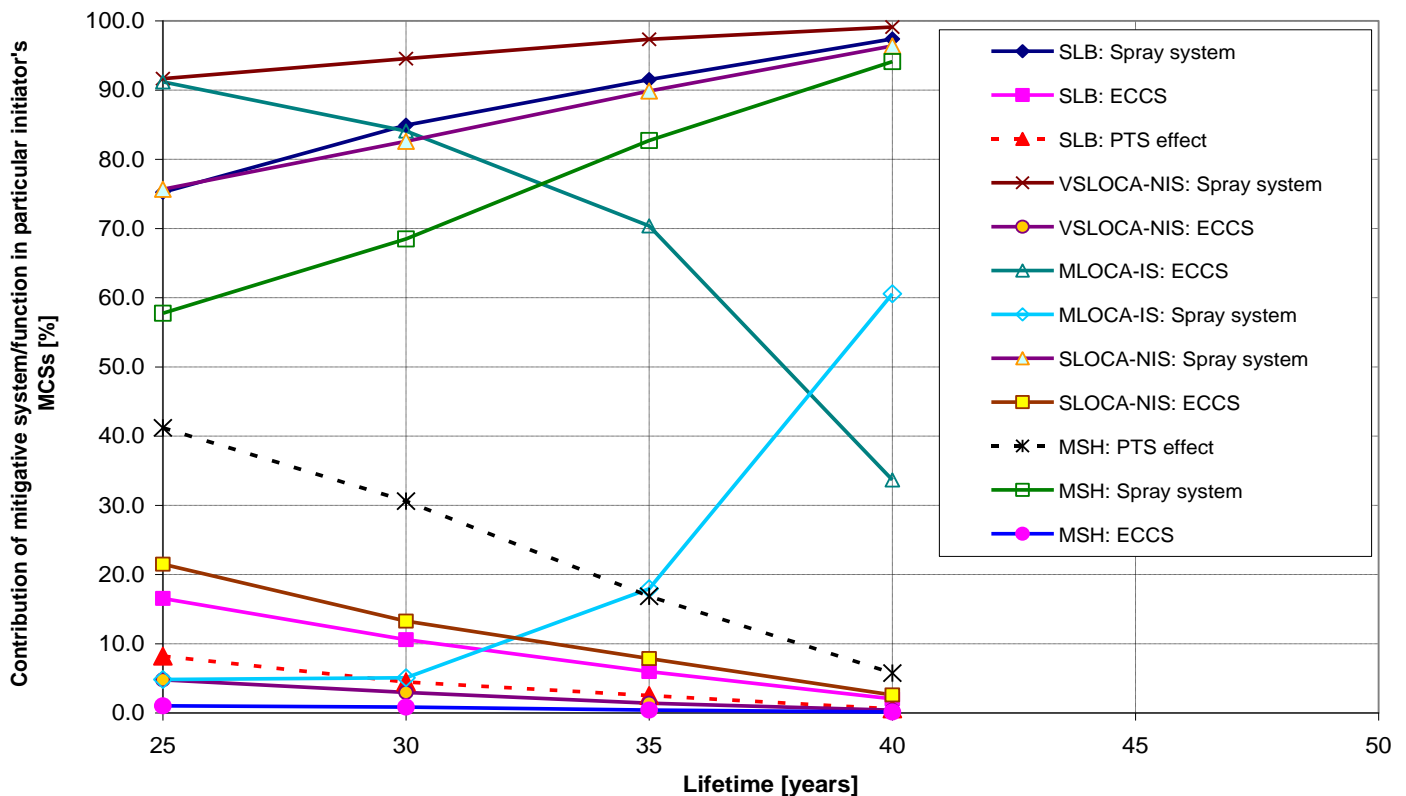


Figure 6 - Change of portion of mitigation function in overall MCSs of particular initiator

Figure 3 shows that fraction of ECCS function is decreasing in time as well as portion coming from support systems failure and PTS effect, while portion of risk connected with spray system failures is continuously increasing. Fast steam isolation valves intended to prevent PTS have been installed in 2000, so the ageing trend of these valves was not considered (constant failure rate model was applied for new equipment). It is not the case with ECCS and spray system, hence detailed investigation of those systems reliability behaviour was implemented [8].

³ MLOCA – medium LOCA (100-200mm), SLOCA – small LOCA (32-100mm), VSLOCA – very small LOCA (9-32mm), SLB – steam line break, MSH – rupture of main steam header

Three safety critical cases were selected for time-dependent reliability analysis:

1. **ECCS-4/6 case:** Primary emergency core cooling system success criteria for Large LOCA event (4 pumps out of 6 are necessary)
2. **ECCS-1/6 case:** Primary emergency core cooling system success criteria for Small LOCA event (1 pump out of 6 is necessary)
3. **SS case:** Confinement spray system (with auxiliary function to cooldown ECCS boron tank)

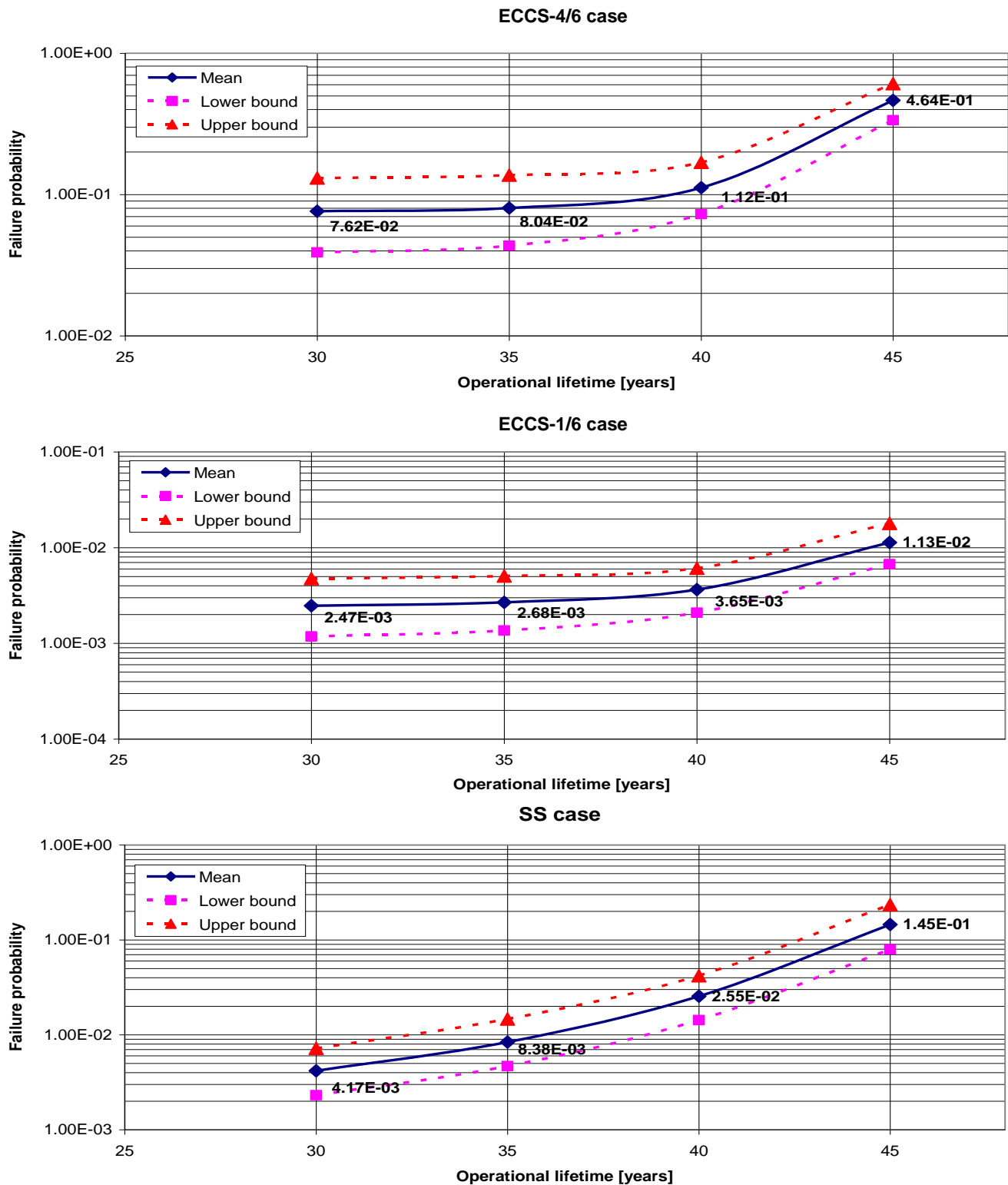


Figure 7 - Change of systems failure probability in time

The cases contain most of the equipments modelled explicitly in current PSA model. Fault trees of selected cases were recalculated for each considered lifetime period. Results of calculations are presented in Figure 7. As it is shown in Figure 7, the failure probabilities of systems are increasing during the lifetime. Rapid increase of failure probability is observed for spray system (by a factor of 35), while for emergency make-up system the rise is much lower (by a factor of 4.5 for ECCS-1/6 case and by a factor of 6 for ECCS-4/6 case). This effect is conditioned by a better redundancy applied in emergency make-up system design, in comparison with spray system.

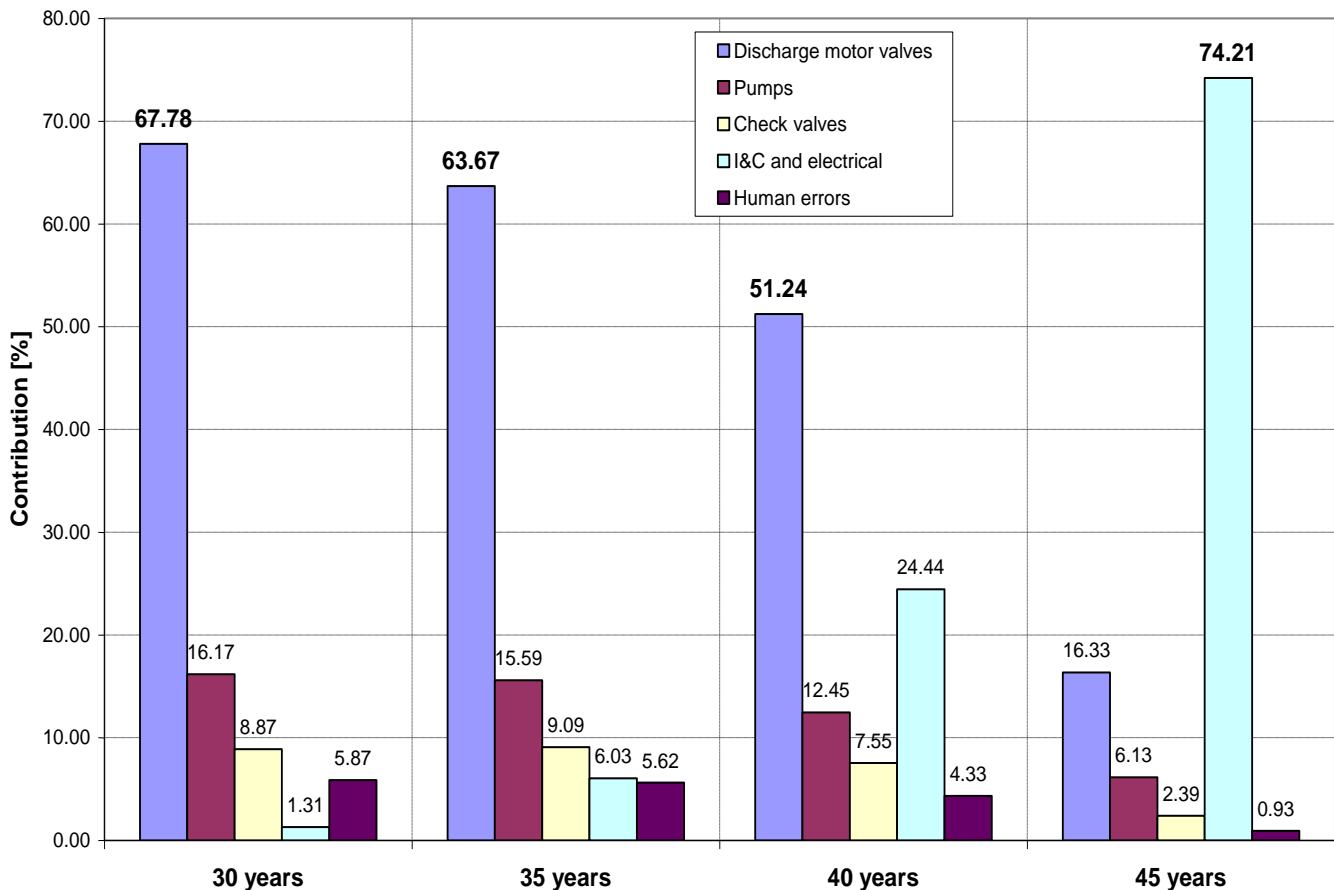


Figure 8 - Contributions of different groups of components to the ECCS failure probability for different lifetime periods

System components were investigated in order to understand the contribution of different group of components to the system failure probabilities. Results of investigation show that the proportions of contribution of different components groups are changing in time (see Figure 8). Particularly significant increase of I&C and electrical equipment contribution is observed. This effect is conditioned by high ageing rate revealed for such components (see table A1.1). However it is necessary to mention that I&C and electrical equipment are easily changeable and observed effect is also driven by assumptions made in the model (for instance see assumption (a)). It may be concluded that systematic monitoring and modification/ replacement of this equipment is necessary in the future.

Importance analysis

Risk importance analysis was performed for all modelled systems, using Fussel-Vesely (FV) factor (see Table 5). The decrease of risk importance for some systems is conditioned by its increase for others, and vice versa (taking into account that risk importance factors are relative).

According to results presented in Table 5 contribution to the overall risk from several systems is rapidly increasing in time (more than a factor of 2 – see orange marked cells). Increasing significance of spray system is already discussed, main reasons of this effect being the LOCA events

sensitivity and problems associated to system design. The rest of orange marked systems are devoted to mitigate consequences of transients, which were found as most vulnerable from ageing point of view.

Table 6 presents the results of importance analysis performed for system components, using Fussel-Vesely factor. Similar components were grouped on generic groups, and the table shows FV parameter for the representative component of the group.

Importance analysis performed at components level shows that importance profile of components is drastically changing due to ageing. Meanwhile this effect is not observed at system level.

Table 5 - Results of risk importance analysis for plant systems

#	System	Type of system	Fussel-Vesely parameter value for analysis cases				X^4
			25	30	35	40	
1.	Spray system	SFS	3.18E-01	3.95E-01	5.65E-01	6.88E-01	↑ 2.17
2.	Emergency core cooling system	SFS	1.36E-01	1.73E-01	1.14E-01	3.71E-02	0.27 ↓
3.	SG diesel make-up system	SRS	1.05E-01	6.66E-02	6.27E-02	9.43E-02	0.90 ↓
4.	AC power supply system	SFS	5.42E-02	5.01E-02	3.40E-02	1.35E-02	0.25 ↓
5.	TG stop valves	SNO	5.26E-02	3.95E-02	2.32E-02	6.88E-03	0.13 ↓
6.	Fast steam isolation valves	SFS	4.39E-02	4.28E-02	4.39E-02	4.62E-02	1.05
7.	Residual heat removal system	SRS	3.09E-02	6.18E-03	1.15E-02	3.50E-02	1.13
8.	Steam dump to the atmosphere	SFS	2.28E-02	2.11E-02	2.74E-02	7.87E-02	↑ 3.46
9.	SG seismic make-up system	SFS	1.99E-02	2.54E-02	4.06E-02	1.20E-01	↑ 6.02
10.	Essential service water system	SFS	1.77E-02	1.93E-02	1.55E-02	6.30E-03	0.36 ↓
11.	Normal primary make-up system	SRS	1.17E-02	1.29E-02	1.48E-02	1.47E-02	1.25
12.	Main feedwater system	SNO	9.09E-03	5.80E-03	3.17E-03	9.19E-04	0.10 ↓
13.	Scram system	SRS	5.49E-03	3.56E-03	1.85E-03	4.87E-04	0.09 ↓
14.	Steam dump to the condenser	SFS	2.95E-03	6.53E-03	1.67E-02	6.01E-02	↑ 20.38
15.	Emergency feedwater system	SFS	1.53E-03	2.55E-03	4.28E-03	2.65E-02	↑ 17.40
16.	Main condensate system	SNO	1.03E-03	1.49E-03	1.38E-03	1.33E-03	1.29
17.	Primary overpressure protection system	SFS	7.94E-04	5.12E-04	2.94E-04	1.07E-04	0.14 ↓
18.	Demineralized water system	SNO	5.56E-04	2.51E-04	5.46E-04	1.48E-03	↑ 2.66
19.	MCP intermediate circuit system	SNO	2.66E-04	2.59E-04	2.93E-04	3.12E-04	1.17
20.	DC power supply system	SFS	1.19E-04	8.76E-05	4.85E-05	1.25E-05	0.11 ↓
21.	Boron preparation system	SRS	1.59E-05	4.01E-06	2.02E-06	8.13E-06	0.51 ↓
22.	Heat sink system	SNO	2.92E-06	5.69E-06	5.80E-06	4.49E-06	1.54

⁴ "X" is a factor which shows difference between FV parameters for 40 and 25 years of operation ($X = FV_{45}/FV_{25}$)

Table 6 - Results of risk importance analysis for components

Component ⁵	System	Fussel-Vesely parameter value for analysis cases				X ⁶
		25	30	35	40	
Boron tank recirculation big valve (2B-20)	SS	1.63E-01	2.10E-01	2.56E-01	2.15E-01	1.31
Confinement spray valves (2B-12/1-3)	SS	1.33E-01	1.69E-01	2.03E-01	1.67E-01	1.26
Diesel-driven pump	DP	8.99E-02	5.62E-02	5.25E-02	7.83E-02	0.87
Boron tank recirculation small valve (2B-20/1)	SS	7.16E-02	8.63E-02	9.69E-02	7.19E-02	1.00
ECCS discharge valves (2B-6,7/1,2)	ECCS	6.09E-02	5.17E-02	3.39E-02	1.10E-02	0.18
Diesel generators	AC	3.54E-02	2.49E-02	1.43E-02	4.20E-03	0.12
TG stop valves relay (ROT)	TG	2.27E-02	1.66E-02	9.78E-03	2.76E-03	0.12
TG stop valves relay (R501)	TG	1.27E-02	9.21E-03	5.43E-03	1.47E-03	0.12
TG stop valves relay (R71)	TG	1.26E-02	9.16E-03	5.40E-03	1.46E-03	0.12
Fast steam isolation valves 1-6	FSIV	1.17E-02	8.97E-03	6.32E-03	3.72E-03	0.32
ESP discharge valves (2AP-5/1,2)	ESFW	9.19E-03	8.25E-03	8.11E-03	8.40E-03	0.91
Diesel pump discharge valve (DP-4)	DP	9.01E-03	8.11E-03	7.67E-03	1.08E-02	1.20
Fast steam isolation valve 7	FSIV	8.63E-03	6.18E-03	3.66E-03	1.11E-03	0.13
ECCS discharge check valves	ECCS	7.67E-03	6.78E-03	4.60E-03	1.52E-03	0.20
NMS discharge main valve 2P-50	NMS	7.41E-03	8.65E-03	1.05E-02	1.13E-02	1.52
ECCS pumps 1-6	ECCS	7.18E-03	5.65E-03	3.51E-03	1.04E-03	0.15
Steam dump valves to atmosphere (BRUA)	SDV	5.82E-03	1.10E-02	1.31E-02	1.16E-02	2.00
Steam pressure sensors (22M-721-726)	FSIV	5.62E-03	4.68E-03	3.79E-03	3.05E-03	0.54
Motor valves on ESWS recirculation (1,2Br-2,6)	ESWS	5.45E-03	5.83E-03	5.26E-03	2.38E-03	0.44
ESP discharge valves (2AP-4/1,2)	ESFW	5.15E-03	5.15E-03	5.66E-03	5.96E-03	1.16
0.4kV safety buses (2804 BNN,2805BNN)	AC	4.52E-03	3.48E-03	2.47E-03	1.40E-03	0.31
Spray system pumps 1-3	SS	3.98E-03	3.08E-03	1.95E-03	6.03E-04	0.15
ESWS pumps (1,2NSO-1-3)	ESWS	3.23E-03	2.37E-03	1.44E-03	3.79E-04	0.12
Steam pressure sensors (21M-721-726)	FSIV	2.56E-03	1.93E-03	1.45E-03	1.05E-03	0.41
Diesel pump discharge valve (DP-2)	DP	2.17E-03	1.80E-03	1.66E-03	2.20E-03	1.01
Spray system discharge valves (2B-10/1,2)	SS	2.05E-03	1.58E-03	9.94E-04	3.06E-04	0.15
Main gate valves of RCS	RCS	1.94E-03	3.20E-03	5.92E-03	8.63E-03	4.45
Scram system relays (RA-5)	Scram	1.90E-03	1.27E-03	7.45E-04	1.77E-04	0.09
Pressurizer level sensors (2R-287)	Scram+ECCS ⁷	1.82E-03	1.23E-03	7.20E-04	1.70E-04	0.09
FSIV Interlock switchers	FSIV	1.77E-03	3.15E-03	3.56E-03	3.63E-03	2.05
ESP pumps (2ASN-1,2)	ESFW	1.70E-03	1.69E-03	1.80E-03	1.60E-03	0.94
ESP discharge valves (2AP-14/2)	ESFW	1.69E-03	1.43E-03	1.40E-03	1.27E-03	0.75
MFW discharge valve (21-26VP-9)	MFW	1.62E-03	1.14E-03	6.41E-04	1.73E-04	0.11
ESP discharge valves (2AP-14/1)	ESFW	1.59E-03	1.34E-03	1.34E-03	1.24E-03	0.78
DG cooling lines valves	AC	1.56E-03	1.05E-03	5.95E-04	1.51E-04	0.10
0.4kV buses (25,26BNN)	AC	1.46E-03	4.15E-03	4.35E-03	2.43E-03	1.67
0.4kV buses (14TR)	AC	1.45E-03	4.13E-03	4.33E-03	2.42E-03	1.67
0.4kV buses (21BN)	AC	1.45E-03	4.13E-03	4.33E-03	2.42E-03	1.67
0.4kV buses (10TR)	AC	1.45E-03	4.13E-03	4.33E-03	2.42E-03	1.67

⁵ Only components important from safety point of view for any of the lifetime periods (25,30,35 or 40) are presented

⁶ "X" is a factor which shows difference between FV parameters for 40 and 25 years of operation ($X = FV_{45}/FV_{25}$). Green marked fields shows cases when relative importance of the component is decreased, orange marked fields represent cases when relative importance of the component increased more than 10 times, white ones – represent the rest of the cases.

⁷ Such kind of sensors provide both signal to scram and activation of one of the safety systems

Component ⁵	System	Fussel-Vesely parameter value for analysis cases				X ⁶
		25	30	35	40	
Primary pressure sensors (AR-650)	Scram+E CCS	1.37E-03	9.10E-04	5.32E-04	1.25E-04	0.09
Scram relays (1RA)	Scram	1.36E-03	9.09E-04	5.32E-04	1.29E-04	0.10
ESP discharge valves (2AP-3/1,2)	ESFW	1.33E-03	1.23E-03	1.42E-03	1.46E-03	1.10
Spray system discharge valves 2B-11/1,2	SS	1.25E-03	9.80E-04	6.32E-04	1.73E-04	0.14
Boron tank (2B-8/2)	ECCS	1.13E-03	8.35E-04	5.02E-04	1.47E-04	0.13
RHR HXs cooling lines valves (TV-5,6)	RHR	8.36E-04	1.18E-03	2.00E-03	3.68E-03	4.40
EFW pumps	EFW	8.36E-04	9.60E-04	1.11E-03	1.18E-03	1.41
RHR pumps inlet valve (2P-14)	RHR	8.32E-04	1.18E-03	1.99E-03	3.66E-03	4.40
NMS pumps cooling lines valves	NMS	7.48E-04	8.75E-04	1.07E-03	1.14E-03	1.53
Switcher of 2B-20 valve	SS	5.44E-04	1.35E-02	5.66E-02	1.64E-01	302
Switchers of 2B-12/1-3 valves	SS	4.74E-04	1.14E-02	4.68E-02	1.30E-01	274
EFW pumps recirculation valves	EFW	3.22E-04	3.92E-04	6.59E-04	1.66E-03	5.15
Boron line discharge valves on NMS inlet (2B-69, 2P-17)	BPS	2.53E-04	1.17E-03	1.50E-03	1.65E-03	6.52
EFW pumps inlet valve (2VP-14)	EFW	2.38E-04	3.36E-04	5.91E-04	1.20E-03	5.06
ESWS recirculation valves	ESWS	1.11E-04	7.51E-05	4.98E-05	9.61E-06	0.09
EFW discharge valves (21,22VP-17)	EFW	5.33E-05	3.40E-04	6.21E-04	1.65E-03	31
FSIVs switchers	FSIV	4.88E-05	1.33E-03	5.21E-03	1.10E-02	226
6kV safety buses (3,4RB-2)	AC	4.83E-05	5.64E-03	5.30E-03	2.49E-03	51
ECCS pumps switchers	ECCS	4.18E-05	8.50E-04	2.58E-03	3.96E-03	94
Switchers of ESP discharge valves 2AP-4/1,2	ESFW	2.72E-05	8.24E-04	4.65E-03	2.67E-02	981
Switchers of steam dump valves to condenser (BRUK)	TG	2.53E-05	7.11E-04	2.98E-03	1.38E-02	546
Switchers of steam dump valves to atmosphere (BRUA)	TG	2.19E-05	9.46E-04	5.80E-03	3.50E-02	1599
Switchers of ESP	ESFW	6.50E-06	4.53E-04	3.20E-03	1.94E-02	2984
Switcher of R-3 regulator (on the line from NMP to Prz)	NMS	3.50E-06	1.28E-04	8.88E-04	3.59E-03	1026
Switcher of EFW discharge valves (21,22VP-17)	EFW	1.45E-06	1.25E-04	1.46E-03	2.16E-02	14845
Switcher of RHR pumps inlet valve (2P-14)	RHR	8.46E-07	4.76E-05	4.17E-04	3.04E-03	3592
Switcher of RHR HXs cooling lines	RHR	8.46E-07	4.76E-05	4.17E-04	3.04E-03	3592
Switcher of RCS main gate valves	RCS	6.54E-07	8.90E-05	8.43E-04	4.57E-03	6988
Interlock switcher of steam dump valves to atmosphere (BRUA)	TG	2.86E-07	9.99E-05	7.87E-04	3.17E-03	11075

Grey shaded cells represent components for which FV value is higher than the commonly used critical criteria 5E-03 [6]. It could be noticed that grey cells are migrating to the bottom of the table in time. Particularly, the following components were not initially covered by FV criteria, but will become critical due to ageing:

- Main gate valves of RCS,
- Switcher of 2B-20 valve,
- Switchers of 2B-12/1-3 valves,
- FSIVs switchers,
- 6kV safety buses (3,4RB-2),
- Switchers of ESP discharge valves 2AP-4/1,2,
- Switchers of steam dump valves to condenser (BRUK),
- Switchers of steam dump valves to atmosphere (BRUA),
- Switchers of ESP,
- Switcher of R-3 regulator (on the line from NMP to Prz) and
- Switcher of EFW discharge valves (21,22VP-17).

Some of components presented in Table 6 are becoming less important in time, which is conditioned by the following factors:

- Faster increase of failure probability of other systems/ components (consequential increase of their relatively importance) and
- Change of proportion of initiating events contribution in the overall CDF value.

For example, ECCS pumps have a significant role in LOCA related events, for which they are sharing contribution with spray system mitigation function. Observing ECCS pumps, it is clear that reduction of importance is conditioned by the following factors:

- Relatively faster increase of spray system importance in time lead to decrease of ECCS mitigating function role in LOCA contributors (see Appendix 2) and
- Relative increase of steam line breaks contribution portion in time lead to decrease of overall LOCA contribution (see Figure 6).

It could be noticed that components which changed their importance by a factor more than 10, mainly use reliability parameters #1 to #7 presented in table A1.1. Those reliability parameters are calculated using plant-specific and VVER-specific data, [4]. It could be concluded that TIRGALEX data did not have significant influence on final results, which are driven by ageing trend analyses performed based on VVER and plant specific data.

It is necessary to highlight that obtained results are very much dependent on the assumption (a) - no modernizations or significant replacement of equipment are foreseen. Thus appropriate monitoring and renovation is necessary for components #1 to #7 (see table A1.1) where significant ageing trends were found.

Application of PSA results for systems classification

The decision making approach was experimentally applied for safety-related systems classification. Decision making process was applied using PSA information obtained by base case PSA model and Ageing PSA model⁸. The aim of performed analysis was to demonstrate the differences between final decisions made using traditional PSA and Ageing PSA results.

Definition of the task

The aim of analysis was to draw out an outline for test & maintenance strategy at Armenian NPP Unit 2. Current strategy is based on qualitative considerations - all systems are categorized into three categories [14]: safety systems (SFS), normal operating systems related to safety (SRS) and systems dedicated to normal operation (SNO). Test & maintenance strategy, as well as other mandatory requirements are different, depending on system category. However, such kind of categorization does not take into account the quantified risk information. The defined task implies application of PSA results in order to improve current categorization and optimize the applied test & maintenance strategies.

PSA model applicability

Plant-specific PSA models (both base case PSA and ageing PSA) are developed for full power operation and consider the risk of internal initiating events. Application of existing PSA models is limited to the systems which have a certain role in reactor shutdown process, in case of accidents occurred at full power operation.

Probabilistic considerations

⁸ It is assumed that non-probabilistic considerations remain the same for both cases. Hence they were skipped from analysis – example was performed only for probabilistic considerations.

Total of 22 systems are considered (see Table 5 - 11 SFS, 5 SRS and 6 SNO) within the task. Fussell-Vesely risk importance factor has been used as an indicator for system importance⁹. Results of system level risk importance analysis are presented in Table 5. Systems have been grouped based on FV value. Four risk importance groups have been considered:

- 1st group (high importance): $FV > 1E-01$
- 2nd group (medium importance): $1E-01 > FV > 1E-02$
- 3rd group (low importance): $1E-02 > FV > 1E-03$
- 4th group (negligible): $FV < 1E-03$

In cases when FV values were on the edge of the groups criteria, a conservative assignment was done (e.g. system with $FV=9.19E-04$ was assigned to the third group). Summary results of systems grouping is presented in the table 7.

Table 7 - Systems grouping based on risk importance

Group	Base case PSA results		Ageing PSA results ¹⁰	
	System	FV _{BCPSA}	System	FV _{APSA}
1 st group (high)	Spray system	3.18E-01	Spray system	6.88E-01
	Emergency core cooling system	1.36E-01	SG seismic make-up system	1.20E-01
	SG diesel make-up system	1.05E-01	SG diesel make-up system	9.43E-02
2 nd group (medium)	AC power supply system	5.42E-02	Steam dump to the atmosphere	7.87E-02
	TG stop valves	5.26E-02	Steam dump to the condenser	6.01E-02
	Fast steam isolation valves	4.39E-02	Fast steam isolation valves	4.62E-02
	Residual heat removal system	3.09E-02	Emergency core cooling system	3.71E-02
	Steam dump to the atmosphere	2.28E-02	Residual heat removal system	3.50E-02
	SG seismic make-up system	1.99E-02	Emergency feedwater system	2.65E-02
	Essential service water system	1.77E-02	Normal primary make-up system	1.47E-02
	Normal primary make-up system	1.17E-02	AC power supply system	1.35E-02
3 rd group (low)	Main feedwater system	9.09E-03	TG stop valves	6.88E-03
	Scram system	5.49E-03	Essential service water system	6.30E-03
	Steam dump to the condenser	2.95E-03	Demineralized water system	1.48E-03
	Emergency feedwater system	1.53E-03	Main condensate system	1.33E-03
	Main condensate system	1.03E-03	Main feedwater system	9.19E-04
4 th group (negligible)	Primary overpressure protection system	7.94E-04	Scram system	4.87E-04
	Demineralized water system	5.56E-04	MCP intermediate circuit system	3.12E-04
	MCP intermediate circuit system	2.66E-04	Primary overpressure protection system	1.07E-04
	DC power supply system	1.19E-04	DC power supply system	1.25E-05
	Boron preparation system	1.59E-05	Boron preparation system	8.13E-06
	Heat sink system	2.92E-06	Heat sink system	4.49E-06

⁹ Dominant contributor information is more related to initiating events ranking. However it is necessary to highlight that Fussell-Vesely value contains also information on risk impact coming from particular system failure.

¹⁰ 40 years long-term prediction results have been used for comparison purposes in order to capture ageing impact on systems role in plant risk profile.

Highlighted cells in the right column indicate systems for which risk importance group has been changed from lower to higher. Results presented in Table 7 have been used for integrated decision making.

Integrated decision

Integrated decision making was done by combining the current system categorization with the grouping presented in Table 7. Combination of mentioned factors has been done using risk-informed matrix presented in Table 8.

Table 8 - Risk-informed systems zoning matrix

Risk importance Group System type	1st group (high)	2nd group (medium)	3rd group (low)	4th group (negligible)
Safety systems (SFS)	I	II	III	III
Normal operation systems related to safety (SRS)	II	II	III	IV
Systems dedicated to normal operation (SNO)	III	III	IV	IV

Table 9 - Zoning of plant systems for base case and aging PSA results

Base case PSA Zone	System	Ageing PSA Zone
II	AC power supply system	II
IV	Boron preparation system	IV
III	DC power supply system	III
IV	Demineralized water system	IV
I	Emergency core cooling system	II
III	Emergency feedwater system	II
II	Essential service water system	III
II	Fast steam isolation valves	II
IV	Heat sink system	IV
IV	Main condensate system	IV
IV	Main feedwater system	IV
IV	MCP intermediate circuit system	IV
II	Normal primary make-up system	II
III	Primary overpressure protection system	III
II	Residual heat removal system	II
III	Scram system	III
II	SG diesel make-up system	II
II	SG seismic make-up system	I
I	Spray system	I
II	Steam dump to the atmosphere	II
III	Steam dump to the condenser	II
III	TG stop valves	IV

According to the risk matrix, systems are categorized on 4 zones (I, II, III and IV) depending on the type of system and its quantitative risk importance. Each of presented zones implies certain test & maintenance strategy. Graded test & maintenance requirements should be applied for different zones (e.g. most strict requirements should be assigned to the systems in zone I, then zone II, etc.). Development of specific test & maintenance requirements should be done by taking into account the mentioned zoning.

Results presented in Table 7 have been used for systems zoning according to the matrix (see Table 9).

From the results presented in Table 9 it is clear that systems zoning is different for base case PSA model and ageing PSA model application. In particular, the following systems should be assigned to a higher zone (with high requirements), if ageing factor is taken into account:

- Steam dump to the condenser,
- SG seismic make-up system and
- Emergency feedwater system.

For several systems, the assigned zone was decreased due to APSA model application, compared with the base case PSA results (e.g. emergency core cooling system, essential service water system and TG stop valves). In such cases, decision making process could foresee assignment of a higher zone, in order to remain on conservative side. Thereby the main advantage of APSA model application is the possibility to adjust test & maintenance strategies in order to pay more attention on the systems vulnerable to the ageing.

Application of PSA results for modernizations prioritization

Modernization program of ANPP Unit 2 is developed based on complex investigation, which includes both probabilistic and deterministic analysis results. In order to achieve an efficient modernization program, the list of measures to be implemented at the plant requires prioritization.

The decisions on the prioritization also imply consideration of deterministic and probabilistic outputs. In terms of safety-related measures prioritization, the main qualitative reference is the list of safety issues specific for VVER-440 type reactors. Categorization and thorough descriptions of VVER-440 safety issues are presented in IAEA TECDOC-640, [13]. Issues both related to design and operation are ranked, according to their safety significance, in four categories of increasing severity, as follows:

- **Category I:** Issues in Category I reflect a deviation from the recognized international practices. It may be appropriate to address them as part of actions that solve higher priority issues.
- **Category II:** Issues in Category II are of safety concern. Defence in depth is degraded, and actions are required to solve the issue.
- **Category III:** Issues in Category III are of high safety concern. Defence in depth is insufficient. Immediate corrective actions are necessary, and interim measures might also be necessary.
- **Category IV:** Issues in Category IV are of critical concern. Defence in depth is unacceptable. Immediate action is required to overcome the issue. Compensatory measures have to be established until the safety problems are solved.

PSA output implies information on dominant initiating events (IE) contribution (see Figures 2 and Figure 3) and risk importance parameters for systems (see table 5). For PSA output in terms of contribution from different IE categories, four contribution groups have been considered:

- 1st group (high importance): CDF^{IE} (contribution from particular IE category) > 30% of overall CDF
- 2nd group (medium importance): 30% of total CDF > CDF^{IE} > 20% of total CDF
- 3rd group (low importance): 20% of total CDF > CDF^{IE} > 10% of total CDF
- 4th group (negligible): CDF^{IE} < 10% of total CDF

The combination of TECDOC-640 categorization and PSA information is presented in table 10.

Table 10 - Risk-informed prioritization matrix

PSA output TECDOC-640 issue	1 st group (high)	2 nd group (medium)	3 rd group (low)	4 th group (negligible)
IV category safety issue	I	II (a)	III (a)	III (b)
III category safety issue	II (a)	II (b)	III (b)	IV
II and I category safety issue	III (a)	III (b)	IV	IV

In comparison with the matrix presented in Table 8, here the intent was to assign more detailed priorities for safety-related measures. That is why the prioritization was done also for items within particular zone (e.g. II(a) is considered to have higher priority than II(b)).

Table 11 presents information on outputs from TECDOC-640 and PSA, and the corresponding measure priority assigned, based on matrix presented in table 10.

Table 11 - Summary information of TECDOC and PSA outputs, in terms of measures planned at ANPP

Measure	Relevant IE category	TECDOC – 640 output	BASE CASE PSA				AGEING PSA			
			PSA ranking			P ¹¹	PSA ranking			P
			IE ¹²	RI ¹³	PSA output ¹¹		IE	RI	PSA output	
Complex investigation of reactor pressure vessel capacity	LOCA	1	1 (45.37%) ¹⁴	-	1	I	1 (38.33%)	-	1	I
Modification of the spray system	LOCA	1	1 (45.37%)	1	1	I	1 (38.33%)	1	1	I
Modernization against sump clogging effect	LOCA	1	1 (45.37%)	1	1	I	1 (38.33%)	1	1	I
Evaluation of essential service water system pipeline breaks in boron room	LOCA	1	1 (45.37%)	1	1	I	1 (38.33%)	1	1	I
Modernization of ECCS	LOCA	1	1 (45.37%)	1	1	I	1 (38.33%)	2	1	I
Confinement safety valves reliability enhancement	LOCA	2	1 (45.37%)	-	1	II(a)	1 (38.33%)	-	1	II(a)
Improvement of confinement leak-tightness	LOCA	2	1 (45.37%)	-	1	II(a)	1 (38.33%)	-	1	II(a)
Installation of remote shutdown panel	LOCA	2	1 (45.37%)	-	1	II(a)	1 (38.33%)	-	1	II(a)

¹¹ Each measure could be related both to the IE contribution and specific system. In such cases, a higher group was assigned as PSA output information (e.g. for Reliability enhancement of RHR system the IE=3 and RI=2, the final PSA output is considered 2).

¹² IE – group assigned based on initiating event contribution

¹³ RI - group assigned based on system risk importance

¹⁴ Value in brackets reflects portion of particular IE contribution in overall CDF

Measure	Relevant IE category	TECDOC – 640 output	BASE CASE PSA				AGEING PSA			
			PSA ranking			P ¹¹	PSA ranking			P
			IE ¹²	RI ₁₃	PSA output ₁₁		IE	RI	PSA output	
Verification of the capability of ventilation system for cooling ECCS and spray system after reconstruction	LOCA	2	¹ (45.37%)	1	1	II(a)	¹ (38.33%)	1	1	II(a)
Verification of the capability of ventilation system for cooling of reliable power supply switchgear	-	2	-	2	2	II(b)	-	2	2	II(b)
Reliability enhancement of RHR system	Transient	2	³ (10.78%)	2	2	II(b)	³ (13.91%)	2	2	II(b)
Complex PTS evaluation	SLB	2	² (25.55%)	2	2	II(b)	¹ (39.75%)	2	1	II(a)
Modification of emergency feedwater system	Transient	1	³ (10.78%)	3	3	III(a)	³ (13.91%)	2	2	II(a)
Modification of the primary circuit overpressure protection system	Transient	2	³ (10.78%)	4	3	III(b)	³ (13.91%)	4	3	III(b)
Secondary side piping reliability assessment	SLB	3	² (25.55%)	2	2	III(b)	¹ (39.75%)	2	1	III(a)
Investigation of external grid recovery action in case of LOSP	LOSP	3	⁴ (1.23%)	-	4	IV	⁴ (0.28%)	-	4	IV
Enhancement of reactor protection system reliability	Reactivity accidents	3	⁴ (2.26%)	3	3	IV	⁴ (2.29%)	3	3	IV

From Table11 results it could be concluded that the overall prioritization profile is quite similar for base case PSA and ageing PSA models. However some differences still exists, for instance the measure related to “Modification of emergency feedwater system” has changed it priority from III to II. Also sub-priorities of “PTS evaluation” and “Secondary side piping reliability assessment” have been changed with APSA model.

It could be concluded that although APSA model could not significantly change the prioritization of modernizations.

4. CONCLUSIONS

Time-dependent reliability analysis has been performed for selected equipments, modelled in PSA of Armenian NPP Unit 2. Increasing ageing trends have been found and trend patterns were identified for several components, particularly for 6kV buses, switcher, 6kV/0.4kV breakers, motor valves and pressure sensors.

Evaluated reliability parameters have been integrated in PSA model which was used for recalculations for different lifetime periods (25, 30, 35, 40 and 45 years of operation). The results of quantification show that CDF value is significantly increasing in time. In particular, a huge increase of CDF value (by a factor of 120) was observed in the period from 40 to 45 years of operation, but large uncertainties found for long-term predictions (45 years operation). It has been concluded that adequate predictions could be done only for relatively short time periods – comparable with already collected statistics. Meanwhile it is necessary to develop an approach for thorough uncertainty analysis, taking into account the ageing aspects.

Quantifications were implemented for base case PSA and ageing PSA models. The obtained results have been compared in detail, on MCS level, system level and on components level.

MCS investigation shows an increasing trend of general transient contribution in overall CDF value. LOCA type events and secondary breaks continue to remain the main contributors in ANPP risk profile.

According to obtained results, the CDF value is highly dependent on the reliability of the following systems and their associated safety functions (conditioned by large contribution of primary and secondary leakage events):

- ECCS to provide cooling water to primary circuit,
- Spray system to provide cooling of boron tank and
- Steam isolation valves to prevent PTS.

Since steam isolation valves are newly installed they were not further analyzed. Detailed investigation was performed to monitor the reliability of ECCS and spray system. Investigation shows that failure probabilities of systems are increasing due to ageing. Rapid increase of failure probability is observed for spray system (by a factor of 35), while for emergency make-up system the increase factor is not more than 6. This effect is induced by a better redundancy applied in emergency make-up system design, in comparison with spray system. Modernisation of spray system is recommended in order to increase its reliability. It is important to mention that the necessity of spray system modernization was also one of the outcomes of base case PSA model quantification. Level of resolution of FT analysis allowed revealing that proportions of contribution of different components groups are changing in time. Particularly, significant increase of I&C and electrical equipment contribution is observed, which is conditioned by high ageing rate observed for such components.

Risk importance analysis was performed for system level and components level using Fussell-Vesely (FV) parameter as an indicator. According to obtained results, the contribution to the overall risk coming from the following systems is rapidly increasing in time (more than a factor of 2).

- Spray system,
- SG seismic make-up system,
- Steam dump systems,
- Emergency feedwater system and
- Demineralized water system.

Significance of spray system was already observed using base case PSA model. Meanwhile importance of the rest of listed systems was found based on Ageing PSA results, and it's induced by

their vulnerability to the ageing process. It is necessary to mention that those systems play an intensive role in the event trees reflecting general transient events, which were found as most sensitive from ageing point of view.

Importance analysis performed at components level shows that importance profile of components is drastically changing due to ageing. In particular, the following components were not important based on base case PSA quantification results, but will become critical due to ageing:

- Main gate valves of RCS,
- Switcher of 2B-20 valve,
- Switchers of 2B-12/1-3 valves,
- FSIVs switchers,
- 6kV safety buses (3,4RB-2),
- Switchers of ESP discharge valves 2AP-4/1,2,
- Switchers of steam dump valves to condenser (BRUK),
- Switchers of steam dump valves to atmosphere (BRUA),
- Switchers of ESP,
- Switcher of R-3 regulator (on the line from NMP to Prz) and
- Switcher of EFW discharge valves (21,22VP-17).

According to obtained results the data used from TIRGALEX databases did not have significant influence on risk profile. Final results of Ageing PSA model are mainly driven by ageing trend analyses performed based on VVER and plant specific data.

It is necessary to highlight that obtained results are very much dependent on the assumption stating that no modernizations or significant replacement of equipment is considered in Ageing PSA model. Thus appropriate monitoring and renovation is necessary for components which are found being vulnerable to ageing.

Principles of integrated risk-informed decision making have been discussed using framework presented in IAEA INSAG-25 document. The proposed decision making approach has been used for:

- plant system classification and
- plant safety-related modernizations prioritization.

The performed exercise allowed demonstrating practical applicability of decision making process and advantages of Ageing PSA model application. It was concluded that Ageing PSA model could be an useful tool for predictive test & maintenance strategy development at system level and useful for improving the modernization program priorities.

It is necessary to stress-out that application of Ageing PSA model allowed to have a broader view to the safety issues for considered NPP. Incorporation of ageing aspects in PSA models shows necessity of modernizations for several systems and components which were hidden in base case PSA model. Having both results, of current situation and prediction of risk profile, it is possible to develop a more precise and efficient action plan for the future.

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APPENDIX 1. TIME-DEPENDENT RELIABILITY DATASET TO BE INCORPORATED IN PSA MODEL

Table A1.1. Components reliability values for selected operational time-points

	Component	Failure mode	Model (pattern)	Base case value in PSA model	Reliability values for selected operational time-points (years)			
					30	35	40	45
1.	6kV buses	Fail to operate	Loglinear	2.13E-07	2.71E-05	4.11E-05	6.22E-05	9.43E-05
2.	Switcher	Fail to operate	Loglinear	1.03E-04	2.59E-03	1.25E-02	6.02E-02	2.90E-01
3.	0.4kV breaker	Fail to change position	Power	2.47E-06	7.83E-05	9.56E-05	1.14E-04	1.33E-04
		Fail to remain in position	Loglinear	5.64E-07	2.29E-06	4.28E-06	8.00E-06	1.50E-05
4.	Motor valve (>100mm)	Fail to change position	Loglinear	7.93E-06	1.09E-05	1.58E-05	2.28E-05	3.31E-05
5.	Pressure sensor	Fail to operate	Loglinear	1.35E-06	1.20E-05	3.91E-05	1.27E-04	4.14E-04
6.	6kV breaker	Fail to change position	Loglinear	4.05E-06	4.21E-06	4.85E-06	5.59E-06	6.45E-06
7.	Motor valve (<100mm)	Fail to change position	Power	3.96E-06	1.55E-05	1.64E-05	1.72E-05	1.79E-05
8.	0.4kV bus	Fail to operate	Linear	4.02E-07	4.09E-07	4.14E-07	4.20E-07	4.25E-07
9.	Manual valve	Fail to change position	Linear	8.69E-05	1.03E-04	1.17E-04	1.31E-04	1.44E-04
		Fail to remain open	Linear	5.86E-07	5.99E-07	6.10E-07	6.21E-07	6.32E-07
		Fail to remain close	Linear	7.01E-08	8.33E-08	9.43E-08	1.05E-07	1.16E-07
10	Intermediate circuit cooling pump	Fail on demand	Linear	3.12E-05	3.25E-05	3.36E-05	3.47E-05	3.58E-05
		Fail to run	Linear	1.23E-04	1.24E-04	1.25E-04	1.27E-04	1.28E-04
11	ESFW pump ¹	Fail on demand	Linear	3.87E-06	5.19E-06	6.29E-06	7.39E-06	8.49E-06
		Fail to run	Linear	9.61E-05	9.74E-05	9.85E-05	9.96E-05	1.01E-04
12	EFW pump ²	Fail on demand	Linear	3.87E-06	5.19E-06	6.29E-06	7.39E-06	8.49E-06
		Fail to run	Linear	9.61E-05	9.74E-05	9.85E-05	9.96E-05	1.01E-04
13	Transformer ≤ 6kV	Fail to run	Linear	7.25E-07	7.35E-07	7.44E-07	7.52E-07	7.61E-07
14	Check valve (D≤100mm)	Fail to open	Linear	8.14E-08	1.04E-07	1.23E-07	1.42E-07	1.61E-07
		Fail to close	Linear	7.32E-07	7.55E-07	7.74E-07	7.93E-07	8.12E-07
15	Check valve (D>100mm)	Fail to open	Linear	2.62E-07	2.85E-07	3.04E-07	3.23E-07	3.42E-07
		Fail to close	Linear	9.30E-08	1.16E-07	1.35E-07	1.54E-07	1.73E-07
		Fail to remain	Linear	1.88E-07	2.11E-07	2.30E-07	2.49E-07	2.68E-07

¹ ESFW – Emergency seismic feedwater

² EFW – Emergency feedwater

	Component	Failure mode	Model (pattern)	Base case value in PSA model	Reliability values for selected operational time-points (years)			
					30	35	40	45
		close						
		Fail to remain open	Linear	9.43E-08	1.17E-07	1.36E-07	1.55E-07	1.74E-07
16	Steam dump valves	Fail to change position	Linear	5.28E-06	6.58E-06	7.67E-06	8.76E-06	9.85E-06
		Fail to remain close	Linear	1.70E-05	2.12E-05	2.47E-05	2.82E-05	3.17E-05
17	Spray pump	Fail on demand	Linear	5.00E-06	5.03E-06	5.06E-06	5.09E-06	5.11E-06
		Fail to run	Linear	2.07E-04	2.08E-04	2.09E-04	2.11E-04	2.12E-04
18	Interlock switcher	Fail to operate	Linear	1.08E-06	2.46E-06	3.61E-06	4.76E-06	5.91E-06
19	Motor driven duct	Fail to change position	Linear	2.64E-05	4.80E-05	6.60E-05	8.40E-05	1.02E-04
20	TG stop valves	Fail to close	Linear	8.46E-06	1.09E-05	1.29E-05	1.49E-05	1.69E-05
21	Emergency make-up pump	Fail on demand	Constant	7.09E-06	7.09E-06	7.09E-06	7.09E-06	7.09E-06
		Fail to run	Constant	2.30E-04	2.30E-04	2.30E-04	2.30E-04	2.30E-04
22	Drainage pumps	Fail on demand	Constant	1.97E-06	1.97E-06	1.97E-06	1.97E-06	1.97E-06
		Fail to run	Constant	1.77E-05	1.77E-05	1.77E-05	1.77E-05	1.77E-05
23	RHR pumps	Fail on demand	Constant	4.17E-06	4.17E-06	4.17E-06	4.17E-06	4.17E-06
		Fail to run	Constant	5,80E-04	5,80E-04	5,80E-04	5,80E-04	5,80E-04
24	Main feedwater pumps	Fail on demand	Constant	2.09E-05	2.09E-05	2.09E-05	2.09E-05	2.09E-05
		Fail to run	Constant	1.75E-05	1.75E-05	1.75E-05	1.75E-05	1.75E-05
25	Diesel-generator	Fail on demand	Constant	1.51E-05	1.51E-05	1.51E-05	1.51E-05	1.51E-05
		Fail to run	Constant	1.82E-03	1.82E-03	1.82E-03	1.82E-03	1.82E-03
26	Primary device of pressure sensors	Fail to operate	Constant	6.49E-07	6.49E-07	6.49E-07	6.49E-07	6.49E-07
27	Fast changing filter	Fail to operate	Constant	8.50E-07	8.50E-07	8.50E-07	8.50E-07	8.50E-07
28	Normal make-up pump	Fail on demand	Constant	5.08E-06	5.08E-06	5.08E-06	5.08E-06	5.08E-06
		Fail to run	Constant	8,70E-04	8,70E-04	8,70E-04	8,70E-04	8,70E-04
29	Primary device of level sensors	Fail to operate	Constant	5.20E-07	5.20E-07	5.20E-07	5.20E-07	5.20E-07
30	Secondary device of level sensors	Fail to operate	Constant	3.70E-06	3.70E-06	3.70E-06	3.70E-06	3.70E-06

Table A1.2. LOCA frequencies values for selected operational time-points

Break size (mm) Years of operation	13	42	77	178	356	788
25 year	6.70E-03	4.70E-04	1.30E-05	1.20E-06	1.10E-07	1.50E-08
30 year	6.10E-03	5.07E-04	1.80E-05	1.73E-06	1.63E-07	2.30E-08
35 year	5.50E-03	5.43E-04	2.30E-05	2.27E-06	2.17E-07	3.10E-08
40 year	4.90E-03	5.80E-04	2.80E-05	2.80E-06	2.70E-07	3.90E-08
45 year	4.30E-03	6.17E-04	3.30E-05	3.33E-06	3.23E-07	4.70E-08

APPENDIX 2. COMPARISON OF MCS ANALYSIS RESULTS FOR BASE CASE AND AGEING PSA MODELS

The table presents contribution from each mitigating function for a particular initiating event group. Values have been calculated for each operational period considered (25, 30, 35 and 40 years).

The table presents only the initiating events which have high contribution to overall CDF (80% contribution) calculated for a particular period, other initiators which had less contribution are not included.

#	Initiating event	MCS groups description	% of CDF _{IE}			
			25	30	35	40
1.	Steam line break	CD due to failure of spray system to cool down boron tank during F&B process (secondary F&B function is failed consequentially by IE - Turbine hall effect)	75.23	84.93	91.51	97.39
		CD due to failure of ECCS to supply water to primary circuit during F&B process (secondary F&B function is failed consequentially by IE - Turbine hall effect)	16.55	10.58	5.97	2.03
		CD due to other failure (mainly COP problem due to FSIV failure)	8.2	4.47	2.51	0.58
2.	RPV rupture	RPV rupture directly leads to CD due to insufficient core cooling capabilities.	100	100	100	100
3.	Unprotected LOCA	Very large LOCAs directly lead to CD due to insufficient core cooling capabilities.	100	100	100	100
4.	Very small LOCA (NIS)	CD due to failure of spray system to cool down boron tank	91.65	94.54	97.34	99.11
		CD due to failure of ECCS to supply water to primary circuit	4.78	2.96	1.40	0.40
		CD due to failure of support systems (ESWS, AC power, etc.)	3.57	2.50	1.25	0.50
5.	Medium LOCA (IS)	CD due to other failure of ECCS to supply water to primary circuit	91.18	84.09	70.38	33.67
		CD due to failure of support systems (ESWS, AC power, etc.)	3.99	10.83	11.70	5.77
		CD due to failure of spray system to cool down boron tank	4.83	5.08	17.93	60.56
6.	Small LOCA (NIS)	CD due to failure of spray system to cool down boron tank	75.66	82.59	89.86	96.38
		CD due to failure of ECCS to supply water to primary circuit	21.49	13.26	7.86	2.59
		CD due to failure of support systems (ESWS, AC power, etc.)	2.85	4.15	2.29	1.03
7.	MSH rupture	COP problem due to FSIVs failure	41.19	30.61	16.86	5.76
		CD due to failure of spray system to cool down boron tank during F&B process (secondary F&B function is failed consequentially by IE - Turbine hall effect)	57.76	68.48	82.71	94.11
		CD due to failure of ECCS to supply water to primary circuit during F&B process (secondary F&B function is failed consequentially by IE - Turbine hall effect)	1.02	0.85	0.41	0.13

#	Initiating event	MCS groups description	% of CDF _{IE}			
			25	30	35	40
8.	Large SGTR	CD due to failure of ECCS to supply water to primary circuit (including failure of ECCS support systems)	95.18	87.58	- ¹⁷	-
		CD due to failure of primary depressurizing function	4.82	12.42	-	-
9.	General transient	CD due to COP effect after failure to close TG stop valves or fail to close steam dump equipment with additional failure of FSIVs	37.69	20.18	11.41	1.55
		CD due to consequential grid failure with failure of DGs and diesel-make-up system.	26.28	20.55	15.39	2.8
		CD due to other failures (failure of secondary heat removal systems)	36.03	59.27	73.2	95.65
10.	Small SGTR	CD due to failure of primary depressurizing function	96.57	80.75	-	-
		CD due to failure of ECCS to supply water to primary circuit (including failure of ECCS support systems)	3.43	19.24	-	-
11.	Large LOCA (IS)	CD due to other failure of ECCS to supply water to primary circuit	82.4	73.28	59.26	-
		CD due to failure of support systems (ESWS, AC power, etc.)	3.49	5.35	6.17	-
		CD due to failure of spray system	14.11	21.37	34.58	-
12.	Small LOCA (IS)	CD due to failure of spray system to cool down boron tank	-	-	80.25	95.7
		CD due to failure of ECCS to supply water to primary circuit	-	-	15.06	3.33
		CD due to failure of support systems (ESWS, AC power, etc.)	-	-	4.69	0.97

¹⁷ “-” mark means that the considered initiating event group is not included in the list of initiating events that contributes with 80% to CDF, for a particular period

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Abstract

The aim of this case study was to demonstrate the benefits of using ageing PSA results in risk-informed decision process.

First stage of the activities was aimed on the development of dataset which reflects age-dependent reliability behaviour for selected components using plant-specific, VVER specific and generic data. The second part describes the integration process of time-dependent reliability dataset in PSA model, quantification and comparison between aging (APSA) and base case PSA decision making results. Practical insights and conclusions are also presented.

The report was prepared by the Nuclear & Radiation Safety Center (NRSC), Yerevan, Armenia, in cooperation with the Institute for Energy and Transport, EC Joint Research Centre, Petten, Netherlands, in the framework of the EC JRC Ageing PSA Network Task 8 activities.

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